50-395

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 8, 1997

Mr. Gary J. Taylor
Vice President, Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
Post Office Box 88
Jenkinsville, South Carolina 29065

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION - INDIVIDUAL PLANT EXAMINATION SUBMITTAL FOR INTERNAL EVENTS (TAC NO. M74475)

Dear Mr. Taylor:

This letter provides the NRC Staff Evaluation Report (Enclosure 1) of the Individual Plant Examination (IPE) for internal events and internal flooding at the Virgil C. Summer Nuclear Station, which was submitted by South Carolina Electric and Gas Company (SCE&G) on June 18, 1993, as supplemented by letters dated February 22, March 20, and August 20, 1996. The NRC staff contractor's Technical Evaluation Report (TER) is also included (Enclosure 2). The enclosed reports are hereby issued to document the staff's findings and conclusions.

A review was performed by the staff which examined the IPE results for their reasonableness considering the design and operation of the Summer plant. The staff employed Brookhaven National Laboratory to review the front-end analysis and back-end analysis of the IPE submittal, and Sandia National Laboratory to review the human reliability analysis (HRA).

The IPE has estimated a core damage frequency (CDF) of 2E-4/reactor-year. Loss of offsite power/station blackouts contribute 39%, transients 25%, loss-of-coolant accidents (LOCAs) 19%, and steam generator tube rupture and interfacing system LOCA, and containment bypasses less than 1%.

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It is the staff's understanding that you used the NUMARC Severe Accident Closure Guidelines to define a vulnerability. Based on this definition, the IPE identified no vulnerabilities. Plant improvements, however, were identified and considered for implementation. Your sensitivity studies (Sections 6.1 and 7, SCE&G IPE Report in Response to Generic Letter 88-20, June 1993) indicate that the revised Loss of Chilled Water Abnormal Operating Procedure alone will reduce the CDF by about 25% to 1.5E-4/reactor-year.

Based on our review of the Summer IPE submittal and associated documentation, we conclude that you have met the intent of Generic Letter 88-20.

Generic Letter 88-20 suggested that licensees use their IPE submittals to address, among other safety issues, Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements," and USI A-17, "Systems Interactions in Nuclear Power Plants." These two issues (Sections 3.3.8, 3.4,

3.4.3, 3.4.4, SCE&G IPE Report in Response to Generic Letter 88-20, June 1993) are adequately resolved for the Virgil C. Summer Nuclear Station.

If you have any questions regarding the enclosed Staff Evaluation Report and/or TER, please do not hesitate to call me on (301) 415-1497.

Sincerely,

(Original Signed By)

Allen R. Johnson, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-395

Enclosures:
1. IPE Staff Evaluation Report
2. Brookhaven IPE Technical Evaluation Report

cc w/enclosures: See next page

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ENCLOSURE 1

SUMMER NUCLEAR POWER PLANT INDIVIDUAL PLANT EXAMINATION

STAFF EVALUATION REPORT

I. INTRODUCTION

On June 18, 1993, South Carolina Electric and Gas Company (licensee) submitted the Summer Individual Plant Examination (IPE) submittal in response to Generic Letter (GL) 88-20 and associated supplements. On January 11, 1996, the staff sent questions to the licensee for more information. On February 22, 1996, the licensee requested additional time to compile, analyze, and submit a comprehensive response. The licensee responded in a letter dated March 20, 1996. On August 20, 1996, the licensee also provided the Summer Nuclear Station Human Reliability Analysis Notebook.

A "Step 1" review of the Summer IPE submittal was performed and involved the efforts of Brookhaven National Laboratory in the front-end and the back-end analyses, and Sandia National Laboratory in the human reliability analysis (HRA). The Step 1 review focused on whether the licensee's method was capable of identifying vulnerabilities. Therefore, the review considered (1) the completeness of the information and (2) the reasonableness of the results given the Summer design, operation, and history. A more detailed review, a "Step 2" review, was not performed for this IPE submittal. Details of the contractor's findings are in the Technical Evaluation Report (Enclosure 2).

In accordance with GL 88-20, the licensee proposed to resolve Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements." and USI A-17, "Systems Interactions in Nuclear Power Plants." No other specific USIs or generic safety issues were proposed for resolution as part of the Summer IPE.

II. EVALUATION

Summer is a Westinghouse 3-loop PWR with a large dry containment. The Summer IPE has estimated a core damage frequency (CDF) of 2E-4/reactor-year from internally initiated events, including the contribution from internal floods. The Summer CDF compares reasonably with that of other Westinghouse 3-loop pressurized-water reactor (PWR) plants. Loss of offsite power (LOOP)/station blackouts (SBOs) contribute 39%, transients 25%, loss of coolant accidents (LOCAs) 19%, and steam generator tube rupture and interfacing system LOCA, and containment bypasses less than 1%. The licensee's Level 1 analysis appeared to have examined the significant initiating events and dominant accident sequences. The common cause factor (CCF) data in the Summer IPE are comparable with that recommended in NUREG/CR-4550. Redundant components were systematically examined to address potential common-cause failures analysis . The approach used was the multiple Greek letter approach (MGL). The methodology followed that described in NUREG/CR-4780 ("Procedure for Treating Common Cause Failures in Safety and Reliability Studies"). The data base used was the EPRI data base (EPRI NP-3967).

Based on the licensee's IPE process used to search for decay heat removal (DHR) vulnerabilities, and review of Summer plant-specific features, the staff finds the licensee's DHR evaluation consistent with the intent of the USI A-45, Decay Heat Removal Reliability, resolution.

Based on the licensee's IPE process used regarding insights to flooding and water intrusion from internal plant sources, the staff finds the licensee's evaluation consistent with the intent of the USI A-17, Systems Interactions, resolution.

The licensee performed an HRA to document and quantify potential failures in human-system interactions and to quantify human-initiated recovery of failure events. Miscalibration events were not explicitly modeled in the analysis of pre-initiator human actions. Therefore the licensee's treatment of miscalibration events may have precluded identification of potentially important pre-initiator events and is considered a weakness of the HRA. The licensee identified the following operator actions as important in the estimate of the CDF:

1. Initiate bleed and feed.

2. Establish condensate feedwater (FW).

3. Establish low-pressure hot leg recirculation.

4. Initiate safety injection and establish emergency FW.

5. Establish low-pressure cold leg recirculation (residual heat removal (RHR) pumps running).

6. Establish low-pressure cold leg recirculation (RHR pumps stopped).

7. Establish high-pressure cold leg recirculation (RHR pumps stopped).

8. Align alternate cooling to charging pumps.

9. Align and start chiller C and pump C to train B.

10. Start second component cooling water (CCW) pump on failure of first pump.

11. For a loss of SW initiator with all the support systems available within two hours, recover at least one train of service water (SW).

12. For a transient initiator (which involves more alarms than that for a loss of SW initiator) with all the support systems available within two hours, recover at least one train of SW.

13. Recover train A of SW when train A failed and only train A support is available within 2 hours (transient initiator).

14. Recover train B of SW when train B failed and only train B support is available within 2 hours (transient initiator).

15. Recover at least one train of chilled water when both trains failed but supported within 2 hours (loss of chilled water initiator).

16. Start chilled water loop B during transient.

17. Start CCW pump B or C during transient.

18. Start train B engineered safety features (ESF) equipment from control board during transient.

The licensee conducted a systematic and detailed HRA of post-initiator human actions. A review of the human error probabilities (HEPs) for all the post-initiator response type actions indicates that in many instances the HEPs tend to be lower than those obtained for similar events HRA number and approach in other IPEs. However, the licensee's consideration of dependencies along with their detailed analysis appears to have resulted in a reasonable ranking of events in terms of their HEPs. The HRA review of the Summer IPE indicated that a viable approach was used in performing the HRA. The submittal appears to meet the intent of GL 88-20 in regards to the HRA.

The licensee evaluated and quantified the results of the severe accident progression through the use of plant response trees (PRTs) that model the plant behavior from the initiating events to end states of fission product releases. The licensee's back-end analysis appeared to have considered important severe accident phenomena. Phenomenological uncertainties were addressed qualitatively in the Summer IPE submittal and were based on technical position papers generated by Fauske and Associates (FAI). According to the licensee, among the Summer conditional containment failure probabilities, early containment failure is negligibly small, late containment failure is 21% with containment overpressurization being the primary contributor, and bypass is negligibly small (0.4%) with steam generator tube rupture (SGTR) being the primary contributor. Also according to the licensee, the containment remains intact 78% of the time. Early radiological releases are dominated by containment bypass sequences and late releases are dominated by transient sequences. The licensee's response to the Containment Performance Improvement Program recommendations is consistent with the intent of GL 88-20 and associated Supplement 3. The overall assessment of the back-end analysis is that the licensee has made reasonable use of back-end techniques in performing the back-end analysis. It must be noted, however, that the PRTs did not include important phenomena which were addressed in the Fauske position papers, such as molten core concrete interactions, and therefore did not consider the effect associated with these phenomena within the PRTs. The exclusion of many containment failure modes from PRT quantification limits the use of the back-end analysis for the investigation of vulnerabilities and of the potential benefit of recovery actions on overall containment performance.

Some insights and unique plant safety features identified at Summer by the licensee are:

1. Summer is designed with two-train redundancy. The SW system, CCW system, chilled water system, and chemical and volume control system each consist of two redundant trains.

2. Summer has numerous mechanisms to dump steam in the event of a plant transient. The steam can be dumped to the condenser via eight valves and to the atmosphere via steam generator power operated relief valves (PORVs) and safety valves (each steam generator has one PORV and five safety valves). During a station blackout, the steam generator PORVs can be operated locally.

3. The emergency FW system has three full-capacity pumps. One of three pumps would meet design heat removal requirements.

4. Two of the three air-operated pressurizer PORVs are supplied with air accumulators. During a loss of instrument air event, these PORVs would still be available.

5. Reactor coolant pump (RCP) seals are cooled by seal injection via the charging system and the thermal barrier cooling via the CCW system. In a LOOP event, the CCW booster pumps, which are supplied by balance-of-plant (BOP) power, would be lost, thereby affecting the cooling of the RCP seals.

6. Summer has a semi-automatic recirculation switchover which automatically opens the containment recirculation sump isolation valves when the refueling water storage tank (RWST) level reaches the low-low setpoint in coincidence with a safety injection signal. However, the operator must close the RWST isolation valves when the sump isolation valves are open.

7. Two of the four reactor building cooling units (RBCUs) will automatically receive SW following a signal to switch to the emergency mode. Only one RBCU is required to provide containment cooling.

8. Summer does not require a fast bus transfer of the BOP power after a reactor/turbine trip.

9. The diesel-driven air compressor, which maintains instrument air, requires no cooling systems because it is air cooled.

10. The anticipated transient without scram (ATWS) mitigation system actuation circuitry provides a backup to the reactor trip system and the engineered safety feature actuation system (ESFAS) for initiating turbine trip and emergency FW.

11. The safety equipment is maintained on a train basis to avoid the loss of both trains due to maintenance.

12. The Summer containment allows water to drain from upper to lower elevations and spill into the reactor cavity. This facilitates debris coolability in the reactor cavity.

The licensee used the NUMARC Severe Accident Closure Guidelines to define a vulnerability. Based on this definition, the IPE identified no vulnerabilities.

The licensee derived several insights regarding potential improvements as a result of the IPE; improvements that were either implemented or under review are as follows:

1. Abnormal operating procedure, "Total Loss of Chilled Water" was developed for establishing alternate cooling for charging pumps to maintain RCP seal injection.

2. A "chiller rotation" policy has been implemented to reduce the downtime of a chiller.

3. Steps were added to an emergency operating procedure (EOP) to monitor diesel generator temperatures and reduce load if temperatures increase.

4. Response to loss of secondary heat sink was made by directing operators to re-energize any pressurizer PORV block valves that were closed and racked out. The steps were moved up in the procedure to allow operators more time to prepare for feed and bleed before a complete loss of heat sink.

5. The turbine-driven FW system pumps were considered to be used to supply FW to the steam generators if the emergency FW system fails. Currently, EOPs call for using FW booster pumps which require steam generator depressurization to less than 305 psig.

6. The computerized "bypasses and inoperable status indication" (BISI) system, which provides a graphic control room indication of critical system operability, was reviewed and updated.

7. Operators are required to re-establish instrument air to the pressurizer PORVs to ensure sufficient air supply is available for multiple openings of the PORVs during feed and bleed.

8. The IPE results have been used to identify drill scenarios that can be used in training and emergency planning.

9. New RCP seal O-rings were considered to be used to provide better performance under loss of thermal barrier cooling and seal injection conditions. This will be phased in during normal pump maintenance starting with the 1997 fall outage.

10. Fire water connection for RCP thermal barrier cooling was considered as an alternate and diverse cooling source for RCP thermal barrier to address loss of RCP seal cooling events.

11. The cooling dependency of the CCW pumps and charging pumps were changed from the chilled water system to the CCW system. The CCW pump motors receive cooling water from the discharge of the CCW pumps (i.e., self-cooled).

12. Key switches were installed to allow use of condensate feed during a loss of emergency FW. Key switches have been provided, with the keys kept in the control room, to bypass FW isolation signals during a loss of heat sink accident.

Based on the information obtained from the accident sequence analysis and the importance ranking, the majority of the sequences are initiated by a LOOP event with a subsequent failure of all onsite power (SBO) or other combinations of system failures that degrade RCP seal cooling and eventually result in a seal LOCA. Small LOCAS with loss of low pressure recirculation also contribute significantly. The dominant failures are associated with the failure of the diesel generators, chilled water chillers, and SW pumps to start and run, and failure to restore offsite power following an SBO.

III. CONCLUSION

Based on the above findings, the staff notes that: (1) the licensee's IPE is complete with regard to the information requested by GL 88-20 (and associated guidance NUREG-1335), and (2) the IPE results are reasonable given the Summer design, operation, and history. As a result, the staff concludes that the licensee's IPE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and therefore, that the Summer IPE has met the intent of GL 88-20.

It should be noted, that the staff's review primarily focused on the licensee's ability to examine Summer for severe accident vulnerabilities. Although certain aspects of the IPE were explored in more detail than others, the review is not intended to validate the accuracy of the licensee's detailed findings (or quantification estimates) that stemmed from the examination. Therefore, this Staff Evaluation Report does not constitute NRC approval or endorsement of any IPE material for purposes other than those associated with meeting the intent of GL 88-20.

ENCLOSURE 2

SUMMER NUCLEAR PLANT INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT